UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 2009

Mr. Robert Grubb
Transnuclear
7135 Minstrel Way, Suite 300
Columbia, MD 21045

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9233, REV. 9, FOR THE MODEL NO. TN-RAM PACKAGE

Dear Mr. Grubb:

As requested by your application dated October 19, 2007, as supplemented September 30, 2008, and February 16, 2009, enclosed is Certificate of Compliance No. 9233, Revision No. 9, for the Model No. TN-RAM package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

Those on the attached list have been registered as users of the package. The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me at (301) 492-3300.

Sincerely,

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9233
TAC No. L24144

Enclosures:
1. Certificate of Compliance No. 9233, Rev. No. 9
2. Safety Evaluation Report
3. Registered Users List

cc w/encls 1 and 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9233 | 9 | 71-9233 | USA/9233/B(U)-96 | 1 | OF | 3 |

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a ISSUED TO *(Name and Address)*

Transnuclear, Inc
7135 Minstrel Way, Suite 300
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. application
dated March 8, 2005, as supplemented.

4 CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: TN-RAM

(2) Description

The package is a steel encased lead shielded cask with wood impact limiters attached at both ends. The cask is a right circular cylinder. The overall dimensions of the packaging are approximately 178 inches long and 92 inches diameter with the impact limiters installed. The cask body is approximately 129 inches long with an outer diameter of 51 inches. The cask cavity has a length of approximately 111 inches and an inside diameter of 35 inches. The cask body is made of a 0.75-inch stainless steel inner shell, a 5.88-inch thick lead annulus, a 1.5-inch thick stainless steel outer shell, a 0.5-inch thick inner bottom plate and a 2.5-inch thick outside bottom plate. The lead shielding is approximately 6 inches thick in the bottom end of the cask. The outer shell of the cask body is covered with a stainless steel thermal shield. The closure lid consists of a 2.5-inch thick outer stainless steel plate and a 0.5-inch thick inner stainless steel plate separated by approximately 6 inches of lead shielding. An optional lid, with the lead shielding in the form of a separate shielding disk, can also be used. The lid is secured by sixteen 1.5-inch diameter closure bolts. Two concentric silicone O-rings are installed in grooves on the underside of the lid. The cask is equipped with a sealed leak test port between the O-rings, a vent port in the closure lid and a sealed drain port in the bottom of the cask. Each impact limiter is attached to the cask by eight 1.75-inch diameter bolts. The cask is equipped with 6 trunnions, four at the top and two at the bottom. The gross weight of the package is approximately 80,000 pounds, including maximum contents of 9,500 pounds.

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with Transnuclear, Inc. Drawing Nos. 990-701, Rev. 8; 990-702, Rev. 7; 990-703, Rev. 9; 990-704, Rev. 5; 990-705, Rev. 6; 990-706, Rev. 4; 990-707, Rev. 4; 990-708, Rev. 7; 990-709, Rev. 2; and 990-710, Rev. 1.

(b) Contents

(1) Type and Form of Material

Dry irradiated and contaminated non-fuel-bearing solid materials contained within a secondary container.

(2) Maximum quantity of material per package

Greater than Type A quantities of radioactive material which may include fissile material provided that the fissile material does not exceed the mass limits of 10 CFR 71.15. The contents may not exceed 1,272 times an $A_2$ quantity. The decay heat of the contents may not exceed 300 watts. The maximum gross weight of the contents, secondary container, and shoring is limited to 9,500 pounds.

6. As appropriate, shoring must be used in the secondary container sufficient to prevent significant movement of the contents under accident conditions.

7. Both the inner cask cavity and the secondary container must be free of water when the package is delivered to a carrier for transport.

8. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Prior to each shipment, the lid seals must be inspected. The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first;

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application; and

(c) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration date: April 30, 2010.

# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Transnuclear, Inc., application dated March 8, 2005.

Supplements dated: May 4, 2007; October 19, 2007; September 30, 2008; and February 16, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: February 23, 2009

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

# SAFETY EVALUATION REPORT
# Model No. TN-RAM Package
# Certificate of Compliance No. 9233
# Revision No. 9

## SUMMARY

By application dated October 19, 2007, as supplemented September 30, 2008, and February 16, 2009, Transnuclear, Inc. (TN) requested an amendment to Certificate of Compliance No. 9233, for the Model No. TN-RAM package. TN requested that the certificate be amended to include the "-96" designation, and that an alternate lid design incorporating a separate shield plug, and an alternate trunnion construction detail, be approved for use. The applicant provided revised packaging drawings with the design changes, an evaluation of the proposed changes, and an evaluation of the package in accordance with 10 CFR 71.19(e).

Based on the statements and representations in the application, as supplemented, the certificate has been amended as requested by the applicant.

## EVALUATION

## 1. GENERAL INFORMATION

### 1.1 Packaging

The TN-RAM is designed for the transport of Type B quantities of radioactive material, primarily in the form of irradiated reactor components, within secondary containers. The cask is constructed primarily of stainless steel with about 6 inches of lead shielding. The cask lid is composed of a 2.5-inch thick stainless steel outer plate and a 0.50-inch thick stainless steel inner plate, with approximately 6 inches of lead shielding between the plates. The applicant requested approval of an alternative design for the cask lid that consists of a 2.5-inch thick stainless steel plate lid, with a separate stainless steel encased lead shield plug. The alternate lid design is the same as the original lid, with respect to bolting and the seal region. The cask body is equipped with a thin stainless steel thermal shield. The gross weight of the package is approximately 80,000 pounds, including the maximum content weight of 9,500 pounds.

### 1.2 Contents

The contents consist of dry, irradiated and contaminated non-fuel-bearing solid materials within secondary containers. The maximum quantity of radioactivity was reduced from 2,000 to 1,272 times an $A_2$ quantity. This change was specified due to changes in the $A_2$ values in 10 CFR Part 71, and is described in Section 5 of this Safety Evaluation Report. The maximum gross weight (9,500 pounds) and the maximum decay heat (300 watts) were not changed. The package may include fissile material provided that the material meets the fissile exempt limits in 10 CFR 71.15.

### 1.3 Drawings

The applicant provided revised packaging drawings that include the optional lid configuration and the optional trunnion construction detail. In addition, minor changes with respect to the design details were included.

The applicant provided the following updated drawings: Transnuclear Drawing Nos. 990-701, Rev. 8; 990-702, Rev. 7; 990-703, Rev. 9; 990-704, Rev. 5; 990-705, Rev. 6; 990-706, Rev. 4; 990-707, Rev. 4; 990-708, Rev. 7; 990-709, Rev. 2; and 990-710, Rev. 1.

## 2. STRUCTURAL

The amendment request proposes three structural design changes: (1) increase of the immersion pressure from 21.0 to 21.7 psig, (2) addition of a "one-piece" trunnion as a design option, and (3) addition of an optional two-piece lid. The increased immersion pressure, per 10 CFR 71.73(c)(6), remains bounded by the original analysis, and is, therefore, acceptable. Revision 4 of Drawing 990-704 notes that the front trunnion can be prepared from a single 15-inch diameter forging, as an option. This change is acceptable because it provides fabrication flexibility without compromising structural capability of the trunnion as originally certified.

The optional two-piece lid is evaluated as follows. The original closure lid, as an integral piece, consists of a 2.5-inch thick outer stainless steel plate and a 0.5-inch thick inner stainless steel plate separated by approximately 6 inches of lead shielding. Drawing 990-710 depicts design details of the optional two-piece lid assembly comprised of a similarly configured 2.5-inch thick stainless steel plate, called the lid flange, and lead shielding in the form of a separate 6.56-inch thick shield disk. The two parts are held in place by sixteen (16) 1-1/2-inch diameter closure bolts in a configuration identical to that for the original closure lid. From a structural performance perspective, due to the absence of a weld between the lid flange and the shield disk that could develop a composite bending resistance, the lid flange must act as the lone load carrying member for the optional lid design.

Section 2.10.8.2 of the application evaluates the structural capability of the optional lid by comparing its strengths, considering factors of safety defined as the ratios of the allowable, and the corresponding calculated stress intensities, to those of the original lid. Figures 2.10.8-1 and 2.10.8-3 depict the ANSYS 2-D axisymmetric finite element models for the original lid and the optional lid, respectively. Since cask internal pressure and all other loads are simulated by pressure equivalents in the ANSYS analyses, the staff agrees with the applicant's approach of evaluating the structural adequacy by examining strength differences between the two lid designs. This is accomplished by comparing stress factors of safety of the original lid to those of the optional lid. Table 2.10.8-1 summarizes stress results and corresponding stress factors of safety for the cask subject to a representative cask internal pressure of 30 psig. For both the primary membrane and primary-plus-bending stress intensity categories, the calculated stresses for the optional lid, which acts without benefit of the composite bending resistance, are higher than those for the original lid, as would be expected. However, considering the optional lid, made of the A-240 Type XM-19 steel for the much higher stress intensity limits than the A-204 Type 304 steel of the original lid, all stress safety factors for the optional lid are shown larger than those for the original lid. This demonstrates acceptable structural performance for the optional two-piece lid.

Section 2.10.8.3 of the application presents the analysis of bolt stresses, in accordance with NUREG/CR-6007, to complement the Section 2.10.1.3 closure bolt evaluation. This includes evaluating the bearing stress on the lid and determining the minimum thread engagement length based on shear stresses in the bolt and flange threads and the associated material strengths. Table 2.10.8-6 lists load combinations and individual stress computation components for five NCT and HAC conditions. The stress results and corresponding stress ratios, which are all less than unity for bolt tensile, shear, bending, and torsional stresses, are summarized in Tables 2.10.8-7 and -8, respectively, and are acceptable.

Section 2.10.9.2 of the application evaluates fatigue performance of the closure lid bolts, assuming that the bolts are replaced after 250 round trip shipments of the TN-RAM package. The evaluation considers the fatigue strength reduction factor of 4.0, per ASME Code, NB-3232.3(c), for stress cyclic histories, such as operating bolt preloads as well as temperature and gasket seating loads. The total fatigue damage factor of 0.89 is calculated to be less than one, and is, therefore, acceptable. Section 8.2.3 notes that the lid, vent, drain, and overpressure transport cover bolts shall be inspected after each use and annually. It also states that the lid bolts shall be replaced at least once per 250 round trips, consistent with the fatigue evaluation results,

The staff has reviewed the packaging structural evaluation and finds that the package meets the requirements of 10 CFR Part 71.

## 3. THERMAL

### 3.1 Discussion

The TN-RAM amendment request proposes the addition of an optional two-piece lid, consisting of a lead shield disc separate from the steel lid. The current lid has integral lead shielding. The thermal analysis results are not sensitive to the differences between the two lid design options because the amount of heat rejected in the axial direction is negligible due to the presence of impact limiters which act as insulators.

In order to update the design to receive a "-96" designation, solar insolation was added to the post-fire ambient condition. The applicant included structural damage due to the free drop and puncture tests. The impact limiters were modified to include the deformation due to the side, corner, and end drops during the fire and post fire analysis, as well as puncture during the post-fire analysis.

The table below provides a summary of component temperatures for the normal conditions of transport (NCT), initial conditions for hypothetical accident conditions (HAC), and HAC analyses conducted by the applicant. The staff noted TN-RAM components maximum temperature increases in the current HAC analysis compared to the original analysis. The applicant stated this was due to the inclusion of insolation during HAC initial conditions, modeling damaged impact limiters during fire and post-fire conditions, an increase in total heat transfer coefficient from the fire to the cask, the application of charring wood temperature at the inner wood surface of the impact limiter, and an increase in outer surface solar absorptivity. All component temperatures remained below the maximum allowable limits.

The lid seal temperature limit provided on pages 3-17 and 4-3 of the SAR is 437°F. This value may be from an outdated reference, the staff referenced the Parker O-Ring Handbook, ORD-5700, Copyright 2007 and Table 3-13 of that reference provided a S0604-70 temperature limit equal to 400°F. The staff also looked at a Parker O-Ring S0604-70 material report from

04/10/1982 which had a recommended temperature limit equal to 450°F. In either case, the HAC lid seal temperature is below the maximum allowable limit.

| Table 3.1 – Summary of Results | | | | |
|---|---|---|---|---|
| Location / Component | Temperature (°F) | | | |
| | NCT | Initial Conditions for HAC | HAC | Maximum Allowable |
| Outer Surface (thermal shield) | 161 | 140 | 1173 | |
| Outer Shell | 164 | 141 | 799 | |
| Lead | 163 | 142 | 610[1] | 621 |
| Inner Shell / Cavity Wall | 163 | 142 | 466 | |
| Lid | 166 | 141 | 496 | |
| Lid Seals | 164 | 141 | 393 | 437 |
| Cavity "Cold Wall" Peak | N/A | N/A | 215[2] | |
| Average Cavity Gas Temperature[3] | 213 | N/A | 516 | |
| Maximum Outer Surface Temperature without Insolation | 103 | N/A | N/A | |

[1] Temperatures reported from behind trunnions.
[2] Peak value of the minimum (coldest) temperature on the cavity wall during accident.
[3] Cavity wall temperature + 50°F.

### 3.4 Thermal Evaluation for Normal Conditions of Transport

The normal conditions of transport results have not been changed from the original analysis. Initial conditions before the thermal accident are established by performing a steady state analysis with a packaging heat load of 300 Watts applied uniformly on the cavity wall surfaces, an ambient temperature of 100°F with solar insolation. The NCT component temperatures are 21°F to 25°F higher compared to the initial conditions before the thermal accident. This is because the analysis was revised to consider 24 hour averaging of solar insolation compared to the more conservative 12 hour averaging, although the initial condition solar absorptivity of the package external surface was increased to 1.0 compared to 0.85 for the original NCT analysis.

### 3.5 Thermal Evaluation for Hypothetical Accident Conditions

The hypothetical accident model was developed using the ANSYS 8.1 computer code. The 3-D model represents the top half of the packaging and includes a 45° sector of the body, lid, trunnions, thermal shield, and impact limiters. Compared to the previous NCT model, impact limiters were added and the dimensions of the lead and steel regions were updated to be consistent with the design. Based on the staff request to revise the application to include structural damage due to free drop, the impact limiters were also modified to include the deformation due to the side, corner, and end drops during the fire and post-fire analyses. The damage modeled was consistent with the structural analysis and accounted for uncertainty in deformation of the impact limiters.

During the thermal accident, heat absorption at the outer surface by radiation and convection is considered. It is assumed that the surface of the packaging is covered with soot during the post-fire conditions and a solar absorptivity of 1.0 for the packaging surfaces is used during the initial conditions and post-fire cool-down period.

Because a puncture may result in tearing of the impact limiter outer steel skin and cause the wood interior of the impact limiter to char, the applicant conservatively assumed the inner surface of the impact limiter inner cover was exposed to a 1112°F char wood temperature for 30 minutes immediately after the end of the fire without heat dissipation from the open surface of the torn wood. This caused a significant increase in seal temperature compared to the previous analysis, but the seal temperature (393°F) remains within the allowable limit. The staff believes the analysis is conservatively bounding due to the location of the application of the maximum char wood temperature, length of time the maximum char temperature was applied, and the fact that the maximum char wood temperature was applied immediately after the fire, when in reality that would not occur until 7.6 hours post-fire. Also, the maximum seal temperature peak would only occur for a few minutes while the maximum allowable temperature limit is based on the long term functional service of 1000 hours.

The model temperature distribution at the end of the fire can be seen in SAR Figure 3.6. The lead temperature reaches a maximum of 610°F at 0.5 hours after the start of the thermal accident. Figure 3.7 of the SAR shows that this occurs in small regions directly beneath the trunnions and the lead temperature is significantly lower at other locations. SAR Figure 3.8 shows the component maximum temperature vs. time during the fire and post-fire periods.

The maximum cask cavity internal pressure during the HAC thermal accident is 28.0 psig which is lower than the MNOP of 30 psig. Previously calculated inner shell, outer shell, and lead temperatures used in the thermal stress analysis are bounding and no revision is required for thermal stress calculations.

On the basis of the review above, the staff concludes that TN-RAM transportation package continues to meet the thermal performance requirements of 10 CFR Part 71 and is suitable for use in accordance with the NRC issued CoC.

## 4. CONTAINMENT

The TN-RAM is a Type B(U) package designed for the transport of dry irradiated and contaminated non-fuel bearing solid materials. A containment review was performed to ensure the amendment request package design was described and evaluated to meet the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The amendment request proposes the addition of an optional two-piece lid, consisting of a lead shield disc separate from the steel lid. The current lid has integral lead shielding. When the two-piece lid is in use, the shield disk is not part of the containment boundary. Seals that form part of the containment boundary include the lid inner silicone o-ring and the vent and drain port silicone o-rings. The seal operating temperature limit is 437°F while the maximum seal temperature is 393°F.

Reactor hardware can include crud surface contamination of which Co-60 is the major radionuclide contributor. The $A_2$ value of Co-60 equal to 11 Ci was used in the analysis. Reference 4-2 of the application states that the maximum measured spot levels on spent fuel rods for various GE BWR plants was 110 – 180 $\mu Ci/cm^2$ at discharge. The number of samples of surface activity reported in Reference 4-2 is very small, and there is a wide range of values. Although NUREG-1617 used a surface activity of 1254 $\mu Ci/cm^2$ which is the maximum value in Reference 4-2 for spent fuel rods, the staff accepted the value of 110 $\mu Ci/cm^2$ used for the analysis as a reasonable estimate for control rod blades.

The applicant calculated the Co-60 crud activity per $cm^2$ based on 1.25 times the smallest maximum spot activity (110 $\mu Ci/cm^2$) which was multiplied by 0.5, assuming the fuel would remain in the reactor and pool for at least one half-life. The applicant then calculated the surface area of a control rod blade (CRB) and based on the typical Co-60 activity per shipment from Section 1.2.3 determined the number of CRBs in a shipment. The Co-60 CRUD activity per $cm^2$ was multiplied by the CRB surface area and the number of CRBs in a shipment to determine the Co-60 crud activity. The CRUD activity was multiplied by the spallation factor of 0.15 to determine the amount of spalled crud that is capable of becoming an aerosol and potentially leaking from containment.

In order to update the TN-RAM design to receive a "-96" designation, the current version of ANSI N14.5 had to be used for leakage rate calculations. The staff evaluated the applicant's leakage rate calculations and determined that ANSI N14.5-97 had been followed. Based on ANSI N14.5 the applicant calculated that the manufacturing and periodic verification tests shall not show a leak rate for the entire containment not greater than $6.87*10^{-4}$ ref*$cm^3$/sec with a test sensitivity of at least $3.4*10^{-3}$ ref*$cm^3$/sec.

The staff agrees that the package meets the containment requirements in 10 CFR Part 71 for a Type B(U) package.

## 5. SHIELDING

The package is designed for the transport of irradiated solids, typically irradiated hardware from power plants. The package was previously evaluated for 2,000 times an $A_2$ quantity of radioactivity. The previous value of $A_2$ for cobalt-60 was 7 curies, and the package shielding was evaluated for 14,000 curies of cobalt-60. Cobalt-60 was chosen as the radionuclide that was most important with respect to shielding considerations. The current value of $A_2$ for cobalt-60 is 11 curies. Therefore the multiple of $A_2$ authorized for transport was reduced to 1,272, to be consistent with the previous shielding analysis.

The applicant requested approval of an alternative design for the closure lid and its lead gamma shielding, and an alternative trunnion construction. The original lid design consisted of a 2.5-inch thick outer stainless plate and a 0.5-inch thick inner stainless steel plate separated by approximately 6 inches of lead shielding. The optional two-piece lid assembly is composed of a 2.5-inch thick stainless steel plate, with a separate lead shield plug that is composed of a 0.375-inch thick stainless steel top plate and a 0.5-inch thick stainless steel bottom plate, with a minimum of 5.68 inches of lead shielding between the plates.

The applicant showed that the optional lid design provides gamma shielding that is bounded by the previous shielding analysis. The minimum lead thickness of the optional lid design (0.68 inches) is the same as was used in the shielding analysis. Therefore no new shielding analyses were performed.

The optional trunnion construction detail does not impact the shielding provided by the packaging.

The staff finds that the revised package design and contents do not affect the ability of the package to meet the external dose rate standards of 10 CFR 71.47. In addition, the package operations specify that radiation surveys are taken prior to each shipment after the contents are loaded.

## 6. CRITICALITY

The package may contain fissile material provided that the fissile material meets the exemption standards in 10 CFR 71.15. Therefore, criticality is not a concern.

## 7. PACKAGE OPERATIONS

The applicant provided revised and updated package operations, including loading with the optional two-part lid, clarifying essential steps in the operations, revising the package leakage testing and acceptance criterion, and clarifying radiation and contamination surveys that are performed prior to each shipment. The staff finds that the revised package operations are adequate.

## 8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant provided revised acceptance tests and a maintenance program for the package. The revised acceptance tests include a load test of the threaded holes which serve as points of attachment for lifting fixtures for the new shield disk, and revisions to the leakage testing for the package. A new Section 8.2.3 was added to describe inspection and maintenance of bolts (lid, vent, drain, and overpressure transport cover bolts). Section 8.2.3 also specifies that lid bolts are replaced at least once per 250 round trip shipments to address bolt fatigue. The staff finds that the changes in the acceptance tests and maintenance program are adequate.

## "-96" EVALUATION

The applicant provided an evaluation of the package to meet 10 CFR Part 71 requirements in accordance with 71.19(e). The applicant provided a revised thermal analysis to address regulatory changes in 10 CFR 71.73(c)(4). The applicant also addressed changes in the $A_1$ and $A_2$ values, which were incorporated into the containment and shielding sections of the application. The staff reviewed the assessment and noted that regulatory changes included in the final rule that became effective April 1, 1996, were not included in the assessment. The staff finds that the changes do not affect the ability of the package to meet the requirements of 71.19(e) for a Type B(U)-96 package.

**CONCLUSIONS**

The Certificate of Compliance has been revised, as requested by the applicant. The following changes were made:

- The Package Identification Number was revised to include the "-96" designation.
- The packaging description in Condition 5(a)(2) was revised to include the optional lid configuration, as well as to clarify the lead shielding dimensions.
- The packaging drawings listed in Condition 5(a)(3) were updated with the new revisions, and the new drawing of the optional lid configuration was included.
- Condition No. 5(b)(2) was revised to limit the total radioactivity to not more than 1,272 times an $A_2$ quantity.

The applicant concluded and the staff finds that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9233,
Revision No. 9 on 2/23/05.